**D. R. Madison (Dennis)**
Vice President - Hatch

**Southern Nuclear**
**Operating Company, Inc.**
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May 3, 2007

SOUTHERN COMPANY
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Docket No.: 50-366

NL-07-0956

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant – Unit 2
Licensee Event Report 2-2007-004
Leak in Reactor Pressure Boundary
Due to Failure of a Socket Weld

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i), Southern Nuclear Operating Company (SNC) is submitting the enclosed Licensee Event Report concerning a reactor pressure boundary leak that resulted from a failed socket weld.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

Dennis Madison

D. R. Madison
Vice President – Hatch
Edwin I. Hatch Nuclear Plant
11028 Hatch Parkway North
Baxley, GA 31513

DRM/MNW/daj

Enclosure: LER 2-2007-004

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. D. H. Jones, Vice President – Engineering
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Hatch
Mr. D. S. Simpkins, Senior Resident Inspector – Hatch

| NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007 |
|---|---|
| **LICENSEE EVENT REPORT (LER)**<br><br>(See reverse for required number of digits/characters for each block) | Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection. |

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**4. TITLE**
Leak in Reactor Pressure Boundary Due To Failure of a Socket Weld

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER(S) 05000 |
| 03 | 09 | 2007 | 2007 | 004 | 00 | 05 | 03 | 2007 | FACILITY NAME | DOCKET NUMBER(S) 05000 |

| 9. OPERATING MODE | 11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 4 | | 20.2201(b) | | 20.2203(a)(3)(i) | | 50.73(a)(2)(i)(C) | | 50.73(a)(2)(vii) |
| | | 20.2201(d) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(ii)(A) | | 50.73(a)(2)(viii)(A) |
| | | 20.2203(a)(1) | | 20.2203(a)(4) | | 50.73(a)(2)(ii)(B) | | 50.73(a)(2)(viii)(B) |
| | | 20.2203(a)(2)(i) | | 50.36(c)(1)(i)(A) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(ix)(A) |
| 10. POWER LEVEL | | 20-2203(a)(2)(ii) | | 50.36(c)(1)(ii)(A) | | 50.73(a)(2)(iv)(A) | | 50.73(a)(2)(x) |
| 000 | | 20-2203(a)(2)(iii) | | 50.36(c)(2) | | 50.73(a)(2)(v)(A) | | 73.71(a)(4) |
| | | 20.2203(a)(2)(iv) | | 50.46(a)(3)(ii) | | 50.73(a)(2)(v)(B) | | 73.71(a)(5) |
| | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i)(A) | | 50.73(a)(2)(v)(C) | | OTHER |
| | | 20.2203(a)(2)(vi) | X | 50.73(a)(2)(i)(B) | | 50.73(a)(2)(v)(D) | | Specify in Abstract below or in NRC Form 366A |

**12. LICENSEE CONTACT FOR THIS LER**

| FACILITY NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Edwin I. Hatch / Kathy Underwood, Performance Analysis Supervisor | 912-537-5931 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX | | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|---|
| B | B21 | N/A | N/A | No | | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|
| | YES (*If yes, complete 15. EXPECTED SUBMISSION DATE*) | NO | | | | |


**16. ABSTRACT** (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On March 9, 2007, at 1230 EST, Unit 2 was in the Cold Shutdown mode. At that time, a leak was identified in a one-inch socket weld elbow, located below the condensing chamber on the "D" Main Steam Line. The Technical Specification (TS) definition of pressure boundary leakage is "leakage through a non-isolable fault in the reactor coolant system." Due to its location, the leak met this definition. Based upon inspection of the weld and adjacent area, it was determined that the leak existed when the Unit was in mode 1. The TS allows no pressure boundary leakage in mode 1.

The cause of the leak is failure of the socket weld due to high cycle fatigue.

Corrective actions for this event included replacing the failed weld and adjacent welds with high cycle fatigue resistant welds and inspection of similar piping to confirm no other leaks.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

## PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

## DESCRIPTION OF EVENT

On March 9, 2007, at 1230 EST, Unit 2 was in the Cold Shutdown mode. At that time, a leak was identified in a one-inch socket weld elbow, located below the condensing chamber on the "D" Main Steam Line (EIIS Code SB). The TS definition of pressure boundary leakage is "leakage through a non-isolable fault in the reactor coolant system." Due to its location, the leak met this definition. Based upon inspection of the weld and adjacent area, it was determined that the leak existed when the Unit was in mode 1. The TS allows no pressure boundary leakage in mode 1. A section of piping was removed and sent to an independent laboratory for analysis. Based on this analysis, the piping was replaced using high cycle fatigue resistant welds (2 to 1 welds).

## CAUSE OF EVENT

The cause of the leak is failure of the socket weld, which was caused by high cycle fatigue.

## REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required, per 10 CFR 50.73(a)(2)(i)(B), because a condition existed which was prohibited by the plant's TS. The Unit 2 TS allows no pressure boundary leakage in mode 1. The long-term increasing trend in drywell floor drain sump in-leakage, the discovery of a leak at a one-inch socket weld elbow located below the condensing chamber on the "D" Main Steam Line, and indication of spray impingement on adjacent components, led to the determination that a leak in the pressure boundary had existed for longer than allowed by the TS. Therefore, the plant was in a condition prohibited by the Unit 2 TS.

The reactor coolant system (RCS) includes systems and components that contain or transport the coolant to or from the reactor core. The pressure-retaining components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. Limits on RCS operational leakage are required to ensure appropriate action is taken before the integrity of the reactor coolant pressure boundary is compromised. The TS delineate the limits on the specific types of leakage.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

The unidentified leakage flow limit allows time for corrective action to be taken before the reactor coolant pressure boundary can be compromised significantly. The five gallons per minute (gpm) limit is a small fraction of the calculated flow from a critical crack in the primary system piping. A critical crack is one large enough to propagate rapidly, ultimately leading to failure of the affected component. Crack behavior from experimental programs shows that leakage rates of hundreds of gallons per minute will precede crack instability (see Unit 2 Final Safety Analysis Report, section 5.2.7.5, "Nuclear System Leakage Detection and Leakage Rate Limits," and Unit 2 TS Bases B 3.4.4, "RCS Operational Leakage").

In this event, unidentified leakage into the drywell had increased linearly from approximately 0.02 gpm at the beginning of March 2006 to approximately 0.13 gpm at the beginning of the refueling outage in February 2007. During outage activities, a leak was identified and investigated. This leak was determined to meet the TS definition of pressure boundary leakage.

At the time the unit was shut down, the unidentified leakage rate was less than three percent of the TS limit of five gpm. The size of the crack was much smaller than the "critical crack" (on which the TS limit is based) as evidenced by the low leakage rate. Therefore, at the time it was discovered and corrective action taken, the crack was not unstable and would not have resulted in catastrophic failure of the line. However, a worst-case instantaneous and complete severing of the one-inch line, due to the presence of a crack, would not result in a significant loss of reactor coolant or present any challenge to core cooling.

A rupture of this one-inch steam line does not result in a significant decrease in water inventory within the vessel. In addition, even if the inventory loss were completely water, the break would still be bounded by both the Loss of Coolant Accident analysis and the Feedwater Line break analysis. This proposed leak is less than 10 percent of the rated capacity of the High Pressure Coolant Injection (EIIS Code BJ) system, which is sized to provide adequate coolant make-up for pipe breaks up to four inches, and approximates the rated capacity of the Reactor Core Isolation Cooling (EIIS Code BN) system. It should be noted that the calculation assumed only liquid flows out of the resulting opening; in reality, a combination of liquid and vapor would flow from the break area. The actual, two-phase flow rate would be lower than that resulting from liquid only. Consequently, either system would have been capable of indefinitely maintaining normal reactor water level. Additionally, a leak of several hundred gpm would have adequately been accommodated by the feedwater system (EIIS Code SJ), which has a flow rate capacity margin at rated conditions of at least 10 percent (over 2000 gpm). Therefore, any one of three diverse and independent high pressure injection systems could have provided sufficient make-up flow to maintain water level well above the top of the active fuel.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

Based upon the preceding analysis, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all operating conditions under which the crack might have propagated to line failure.

## CORRECTIVE ACTIONS

The failed weld and adjacent welds were replaced with high cycle fatigue resistant welds.

Inspection of the three other similar lines was performed. No additional leaks were identified.

## ADDITIONAL INFORMATION

Other Systems Affected: No systems were affected by this event other than those which have already been discussed in this report.

Failed Components Information:
Master Parts List Number: 2B21-Socket Weld
Manufacturer: N/A
Model Number: N/A
Type: N/A
Manufacturer Code: N/A

EIIS System Code: SB
Reportable to EPIX: No
Root Cause Code: B
EIIS Component Code: N/A

Commitment Information: This report does not create any permanent licensing commitments.

Previous Similar Events: No License Event Reports have been reported in the past two years in which a failure of the reactor pressure boundary has occurred.